NON-PUBLIC?: N
ACCESSION #: 8904180201
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Palo Verde Unit 1 PAGE: 1 of 8

DOCKET NUMBER: 05000528

TITLE: Reactor trip Due to Control Element Assembly Calculator Failure
EVENT DATE: 03/05/89 LER #: 89-004-00 REPORT DATE: 04/04/89

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Timothy D. Shriver, Compliance Manager TELEPHONE: 609-393-2521

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: JC COMPONENT: CPC MANUFACTURER: GO80
REPORTABLE TO NPRDS: Y

CAUSE: X SYSTEM: EA COMPONENT: 52 MANUFACTURER: GO80
REPORTABLE TO NPRDS: N

CAUSE: D SYSTEM: EI COMPONENT: 86 MANUFACTURER: X999
REPORTABLE TO NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

March 5, 1989, Palo Verde Unit 1 was operating at approximately 100 percent power when, at approximately 1001 MST a reactor trip occurred on Departure from Nucleate Boiling Ratio (DNBR) due to a Control Element Assembly Calculator (CEAC) failure. During the ensuing turbine/generator trip, power was lost to one of the two 13.8 kv non-class electrical busses when the bus's normal feeder breaker did not trip as required. A small fire was observed and subsequently extinguished on the feeder breaker, trip coil.

The cause of the trip was a failure of the CEAC #2 processor board. The processor board was replaced.

END OF ABSTRACT

I . DESCRIPTION OF WHAT OCCURRED:

A. Initial Conditions:

On March 5, 1989 at approximately 1001 MST Unit 1 was in Mode 1 (POWER OPERATION) at approximately 100 percent power.

B. Reportable Event Description (Including Dates and Approximate Times of Major Occurrences):

Event Classification: An event that results in manual or automatic actuation of the Reactor Protection System.

On March 5, 1989 at approximately 1001 MST, the Palo Verde Unit 1 reactor tripped on Low Departure from Nucleate Boiling Ratio (DNBR).

On March 5, 1989, Unit 1 was operating in a steady state condition at 100% Reactor Power. At approximately 1001 MST, a processor board (CPC) in the Control Element Assembly Calculator (CEAC) #2 (JC) failed. CEAC #2 generated a penalty factor which was then transmitted to the Core Protection Calculators (CPC)(JC). The CPCs compared this information with that provided by CEAC #1 and determined that the information being received from the CEACs deviated more than the allowable tolerances. All four channels of the CPCs then generated a CPC SENSOR FAILURE alarm (IB) and initiated a 10.8 second delay designed to allow time for a spurious deviation to clear. After the delay, all four channels of the CPCs generated a LOW DNBR reactor trip signal resulting in a reactor trip.

The reactor trip precipitated a turbine (TA)/generator (EL) trip and a Fast Bus Transfer (FBT) of the unit loads from the Unit Auxiliary Transformer (UAT)(EL) to the Startup Transformer (EB). The transfer of 13.8 KV bus NAN-SO1 was successful while the transfer of 13.8 KV bus NAN-SO2 was not completed due to the feeder breaker (EA) from the UAT (NAN-SO2A) not tripping. A load shed of NAN-SO2 occurred, resulting in the loss of power (LOP) to several plant components. Plant equipment affected by the LOP included Reactor Coolant Pumps (RCP)(AB) 1B and 2B, the 'A' Heater Drain Pump (SN), the 'C' Condensate Pump (SD), and the 'C' and 'D' Circulating Water Pumps (KE).

When power was lost to the NAN-SO2 bus, the Seismic Monitoring System (IN) alarmed as designed. The seismic unit has an internal battery backup, but the system is not designed to fast transfer electrical power. Thus, the alarm was an expected occurrence.

The loss of power to NAN-SO2 resulted in the Radiation Monitoring System

(RMS)(IL) minicomputer and the New Fuel Area Radiation Monitor (RU-19)(IL) being downpowered. A Radiation Protection

Technician (utility, non-licensed) was stationed at the terminal in the Radiation Protection office to report alarms to the Control Room until the Control Room terminal was reenergized. After power was restored to the components, the RMS minicomputer and RU-19 were restored to service.

The Control Room Operators (Utility, Licensed) entered the Emergency Procedure (41EP-lZZOI) when the reactor trip occurred. The Control Room Supervisor (CRS) (Utility, Licensed) initiated the Diagnostic Flow chart and during its performance noted that the reactor trip first out annunciator panel (IB) did not function as expected. The CRS completed the Diagnostic Flow Chart and correctly diagnosed the event as an uncomplicated reactor trip with a degraded electrical condition, not requiring entry into the Emergency Plan.

At approximately 1010 MST an Auxiliary Operator (AO)(utility, non-licensed) notified the Control Room of a fire in NAN-SO2A (EA) and Fire Protection was subsequently notified. The Reactor Operator (RO) (utility, licensed) attempted to trip the NAN-SO2A breaker but was unsuccessful. The AO then unsuccessfully attempted to trip the breaker using the local trip switch. A second AO (utility, non-licensed) then successfully tripped the breaker by opening the breaker cubicle and using the manual trip lever. The fire was then extinguished by the AOs utilizing the carbon dioxide hose reel in the area. Control Room personnel and Fire Protection were notified that the fire was extinguished at approximately 1016 MST.

The Unit was stabilized in Mode 3 at approximately 1045 MST. Prior to reenergizing the NAN-SO2 bus, a visual inspection and megger test were performed by the Unit Electrical and Protective Relaying and Control (PR&C) groups. With the exception of the NAN-SO2A breaker, no damage was noted. Plant Management authorized the reenergization with concurrence from the Unit Electrical, Unit Operations, and the Engineering Evaluations Department personnel. At approximately
522 MST the NAN-SO2 bus (EA) was reenergized.

C. Status of structures, systems, or components that were inoperable at the start of the event that contributed to the event:

There were no structures, systems, or components inoperable prior to this event which contributed to this event.

D. Cause of each component or system failure, if known:

The cause of the failed CEAC (JC) was the failure of its processor board to correctly execute several instructions. The root cause of the failed processor board has been isolated to the failure of one of the arithmetic logic unit integrated circuits.

The root cause has not been conclusively determined for NAN-SO2A failing to trip when required. The most probable causes are either a slight armature linkage misalignment, trip coil degradation, or the lack of a general breaker overhaul.

E. Failure mode, mechanism, and effect of each failed component, if known:

The failure of the processor board to correctly execute several instructions caused erroneous calculation of Penalty factor based on non-deviating CEA positions. The CEAC 2 Penalty factor caused the CPCs to generate a low DNBR trip signal which tripped the reactor.

In the case of the NAN-SO2 breaker, the trip coil energized as designed, but the slug failed to be drawn into the trip coil. Thus, the latching mechanism failed to release the spring to open the breaker. Additionally, since the breaker did not open, a contact designed to deenergize the trip coil when the breaker opens, remained shut by design and allowed the trip coil to overheat causing the fire.

F. For failures of components with multiple functions, list of systems or secondary functions that were also affected:

Not applicable - The components described in Section I.D are single function components.

G. For failures that rendered a train of a safety system inoperable, estimated time elapsed from the discovery of the failure until the train was returned to service:

No train of safety systems was rendered inoperable.

H. Method of discovery of each component or system failure or procedural error:

The CEAC processor board failure was discovered during troubleshooting in accordance with an approved work document. The NAN-SO2A breaker malfunction was discovered during troubleshooting in accordance with an approved work document.

I. Cause of Event:

The Reactor trip was caused by a failure of a processor board in CEAC #2 as discussed in Section I.D. This represents an isolated component failure.

The NAN-SO2A breaker malfunction cause has not been conclusively determined as described in Section I.D.

J. Safety System Response:

In this event the reactor protection system (JC) operated as designed upon detecting the CEAC-generated penalty factors. No other safety systems were called on to function.

K. Failed Component Information:

Processor Board on CEAC #2 - Interdata Model 35-659 NAN-SO2A Breaker - General Electric Magnablast AM 13.8-1000-4H

II. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS EVENT:

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of design bases anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1860 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

C. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the calculated core DNBR is sufficiently greater than 1.24 such that the

decrease in calculated core DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modeling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays. One of the routines performed by CPC's utilizes penalty factors from the CEAC's when CEA deviations exist.

The reactor trip occurred due to all four Core Protection Calculators (CPC) calculating a DNBR value exceeding the setpoint. The CPC calculations resulted from an abnormally large penalty factor being generated in Control Element Assembly Calculator (CEAC) #2 due to a component failure. Thus, the actual DNBR safety limit was not exceeded.

There were no other Engineered Safety Features Actuation System (ESFAS) actuations during the event and none were required.

The reactor was stabilized in Mode 3 with feedwater provided initially by the reactor trip override (RTO) circuit and then single element control of the Feedwater Control System (FWCS). SG levels exhibited the expected response throughout the event and no safety limits were challenged. All eight Steam Bypass Control System (SBCS) valves received an initial quick open demand then closed as the excess stored energy in the Steam Generators (SG) and main steam piping was released. SBCS then provided SG pressure control and Reactor Coolant System temperature control with the modulation of SBCS valve PV-1001. The SBCS performed as designed.

RCP's (AB) 1B and 2B lost power at 1001 MST due to their power supply NAN-SO2, failing to fast transfer to its alternate supply when the Unit Auxiliary Transformer was tripped. The PVNGS safety analysis assumes a loss of forced circulation involving the loss of all four RCP's from 100 percent power. The existing situation with a loss of two pumps from a shutdown condition is well within the bounds of the safety analysis.

The loss of RU-19, which monitors radiation levels in the new fuel storage area, is of no safety consequence as the Fuel Building effluent monitor (RU-145) was still in service and there was no fuel in the new fuel storage racks.

In summary, the required safety systems performed as designed. Normal operation of non-class equipment maintained control of safety functions and successfully controlled the event from initiation through plant stabilization. Thus, this event did not impact the health and safety of the public.

III. CORRECTIVE ACTIONS:

A. Immediate:

Operator action was taken to place the reactor plant in a stable condition in accordance with the appropriate operating instructions. Operator actions stabilized the reactor plant at approximately 1045 MST and restored power to the deenergized NAN-SO2 bus at approximately 1522 MST. The following equipment was placed in quarantine to minimize the loss of information: CEAC #1, #2; CPC's B, C, D; the NAN-SO2 breaker panel, and the Control Element Drive Motor Control System.

Extensive troubleshooting efforts were successful in locating the problem in the CEAC (JC) and the processor circuit board was replaced. The processor board failure has been determined an isolated occurrence and further corrective action is not deemed warranted at this time.

The trip coil for the normal feeder breaker NAN-SO2A (EA) was replaced. In addition, preventive maintenance was performed on the breaker in accordance with the vendor technical manual.

B. Action to Prevent Recurrence:

With regard to the NAN-SO2A breaker, the following actions will be taken:

1) The preventive maintenance procedures and tasks applicable to all three units will be revised based upon an engineering evaluation of the Technical Manual and additional information recently provided by the vendor. These changes are expected to be complete by April 30, 1989.

2) Procedural controls governing the PM program will be revised to require technical justification and Unit Maintenance Management approval for waiving a PM task. This approval to waive a PM task will be escalated on successive waivings. This control is expected to be proceduralized by May 31, 1989. The Plant Director (utility, non-licensed) has issued a memo to implement this policy as an interim action until the revised procedural controls become

effective.

3) Verification of breaker operation by cycling will be conducted for the following 13.8 kv breakers:

- Normal supply breakers NAN-SOlA and NAN-
O2A

- The RCP breakers

- The alternate supply breakers NAN-SOlB and NAN-SO2B.

The complete cycling of these breakers is expected to occur prior to reaching 30% reactor power upon restart.

4) Although the overheating of the trip coil was an expected response to the NAN-SO2A breaker not opening, an attempt will be made to determine if the trip coil was degraded prior to the event. Contractual agreements are being formulated with the vendor to perform destructive examination of the trip coil. The results of this examination are expected by June 30, 1989. Since this item is dependent on the vendor scheduling, the cycling of breakers addressed in Section III.B.3 will provide assurance of the trip coils ability to perform their design function.

IV. PREVIOUS SIMILAR EVENTS:

Although previous trips have occurred on indication of exceeding the Departure from Nucleate Boiling Ratio (DNBR) setpoint, none of these arose from the circumstances reported herein. Thus, there are no previous similar events.

ATTACHMENT 1 TO 8904180201 PAGE 1 OF 1

Arizona Nuclear Power Project

P.O.BOX 52034 PHOENIX, ARIZONA 85072-2034
192-00463-JGH/TDS/RJR
April 4, 1989

U. S. Nuclear Regulatory Commission
NRC Document Control Desk
Washington, D.C. 20555

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 1
Docket No. STN 50-528 (License No. NPF-41)
Licensee Event Report 89-004-00

File: 89-020-404

Attached please find Licensee Event Report (LER) No. 89-004-00 prepared and submitted pursuant to 10CFR 50.73. In accordance with 10CFR 50.73(d), we are herewith forwarding a copy of the LER to the Regional Administrator of the Region V office.

If you have any questions, please contact T.D. Shiver, compliance Manager at (602) 393-2521.

Very truly yours,

J. G. Haynes
Vice President
Nuclear Production

JGH/TDS/RJR/kj

Attachment

cc: D. B. Karner (all w/a)
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